Department of Energy
Washington, D.C. 20545
Docket No. 50-537
HQ:S:83:182

JAN 11 1983

Mr. Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Check:

ADDITIONAL INFORMATION ON MECHANICAL ENGINEERING BRANCH (MEB) ITEMS 4, 26, 64, 68, 69, AND 72

References: Letter HQ:S:82:143, J. R. Longenecker to P. S. Check, "Meeting Summary: November 22-24, 1982, Mechanical Engineering Branch/Clinch River Breeder Reactor Plant Meeting," dated December 14, 1982

Letter HQ:S:82:128, J. R. Longenecker to P. S. Check, "Additional Information Resulting from the September 8-9, 1982, MEB/CRBRP Meeting," dated November 23, 1982

Letter HQ:S:83:181, J. R. Longenecker to P. S. Check, "Additional Information on Steam Generator Non-Destructive Examinations (NDE) and Reactor Vessel (RV) Core Support Cone Structural Integrity," dated January 11, 1983

Enclosed is additional information concerning MEB items 4, 26, 64, 68, 69, and 72 from Reference 1.

The response to MEB question 4, previously submitted in Reference 2, has been revised in response to comments from EG&G, Idaho Falls, and is enclosed.

Responses to MEB questions 26, 64, 68, and 69 are enclosed to complete actions previously committed to in Reference 2. A response to MEB item 72 has been provided under separate cover in Reference 3. The enclosed modifications to the Preliminary Safety Analysis Report will be included in a future amendment.

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Questions regarding the enclosure may be addressed to Mr. D. Robinson (FTS 626-6098), Mr. D. Hornstra (FTS 626-6110), or Mr. D. Edmonds (FTS 626-6157) of the Project Office Oak Ridge staff.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of
Breeder Demonstration Projects
Office of Nuclear Energy

Enclosure

cc: Service List
Standard Distribution
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Item 4 NRC Question:

Do any mechanical systems and components correspond to Quality Group D requirements as contained in Regulatory Guide 1.26? (Item 2 pg. 3.2.2-3)

Response:

A separate category of equipment equivalent to Quality Group D has not been specified for CRBRP. However, any CRBRP equipment which are equivalent to equipment covered in Quality Group D of Regulatory Guide 1.26, have quality requirements corresponding to Reg. Guide 1.26. This is accomplished by the CRBRP Quality Assurance Program, as discussed in PSAR Chapter 17, Appendix A, Section 0.3 and the imposition of appropriate industry standards.

A listing of industry standards being applied to non-safety related equipment is provided in PSAR Table 3.2-4.

MEB Item 26: The NRC expressed concern at the November 22-24, 1982, meeting at Waltz Mill that no specific criterion was identified for the evaluation of flow induced vibration (FIV) test results. It was suggested that a limiting value of 50 percent of the Code endurance limit at 10^6 cycles would be appropriate.

Response: The information presented at the same meeting for MEB Item 64 indicated that for load controlled conditions the high cycle loadings for CRBRP require evaluation at about 10^9 to 10^{10} cycles. Since the endurance limit decreases by approximately a factor of 2 in going from 10^6 to 10^9 cycles, the CRBRP procedures are equivalent to the suggested limiting value. In any event, the FIV results must be within the component design limits or corrective action will be required as noted in PSAR Section 3.9.1.

MEB Item 64, Part 1: The Applicant used modified creep-fatigue damage rules for non-Code stamped austentic stainless steel components. The modified rules assumed that in compressive hold, the creep damage is only 20% as damaging as that caused by the same sustained stress in tension. Other studies indicate that this may not be conservative and so the Applicant should justify the 20% factor.

Response: Technical justification for the modified compressive hold rules was provided at the November 22-24, 1982, meeting with the NRC MEB at Waltz Mill, Pennsylvania (see attachments to DOE letter HQ:S:82:143 dated December 14, 1982). The enclosed changes to PSAR Section 4.2 document the procedure used to perform creep damage calculations. Appendix 5.2A is deleted because it duplicates the description of the modified rules given in Section 4.2.

REPLACE WITH INSERT 4.2-181

(7) Subject to the above limitations, the creep damage may be calculated in accordance with F9-4T and Code Case 1592 as modified. The modification is to use a peak stress to rupture design curve based upon the stress to rupture design curve in Code Case 1592 adjusted for the influence of a non-linear stress state caused by the presence of a geometric stress concentration as with the following:

Step 1 - Determine the smooth speciment stress rupture strength curve by tests of the same material at the temperature of interest.

Step 2 - Determine the stress rupture strength curve with the presence of the geometric stress concentrations under the same conditions in (1) with specimens of the same heat of material with the same histories. Analytically determine the peak stress relative to the net stress thus defining the stress rupture strength in terms of "peak stress" vs. time to rupture.

Step 3 - Ratio Code Case 1592 stress to rupture design curve by the ratio of Step 1 divided by Step 2. This must be done for at least 3 points in time with a separation in time of at least two orders of magnitude. In cases where the strength ratio varies with lifetime, the lesser of the value extrapolated to the component lifetime or the experimental value for the longest duration tests shall be used.

(8) The total creep-fatigue damage is determined by adding to the creep damage and fatigue damage calculated in accordance with T-1411, -1412, -1413, and -1414 of Code Case 1592.

(9) The allowable creep-fatigue damage (D) is determined from the lesser of the values from Figure T-1420-2 of Code Case 1592 (See Figure 4.2-47a) and an average of test values from creep-fatigue interaction tests of notched specimens.

~~(10) The greater of the damage using the modified rule and the damage using the stress unaltered by the stress concentrations and the Code Case 1592 stress to rupture design curve shall be used.~~ 49

High Cycle Strain Controlled Fatigue Limits

For those 304 and 316 Stainless Steel components which are outside ASME Code jurisdiction, the fatigue damage for strain controlled cyclic deformations in excess of $1 \cdot 10^6$ cycles may be evaluated using allowable strain ranges obtained from Figure 4.2-47B, provided metal temperatures do not exceed 1100^oF. Fatigue life reduction factors must be applied independently for slow strain rates and hold times, in accordance with ASME Code requirements. 57

(7) Subject to the above limitations, the creep damage may be calculated in accordance with F9-4T and Code Case 1592 with one of the following modifications.

(A) Use a peak stress-to-rupture design curve based upon the stress-to-rupture design curve in Code Case 1592 adjusted for the influence of a non-linear stress state caused by the presence of a geometric stress concentration.

(a) Determine the smooth specimen stress rupture strength curve by tests of the same material at the temperature of interest.

(b) Determine the stress rupture strength curve with the presence of the geometric stress concentrations under the same conditions in (a) with specimens of the same heat of material with the same histories. Analytically determine the peak stress relative to the net stress thus defining the stress rupture strength in terms of "peak stress" vs time to rupture.

(c) Ratio the Code Case 1592 stress to rupture design curve by the ratio of (b) divided by (a). This must be done for at least 3 points in time with a separation in time of at least two orders of magnitude. In cases where the strength ratio varies with lifetime, the lesser of the value extrapolated to the component lifetime or the experimental value for the longest duration tests shall be used.

(d) Use the greater of the creep damage using this modified rule and the creep damage using the stress unaltered by the stress concentration and the Code Case 1592 stress-to-rupture design curve.

(B) If tests subject to the above limitations (1 through 6) show no decrease in rupture life for prototypic notch geometries, calculate the component creep damage neglecting the stress concentration due to the notch. No reduction in damage below the damage using the stress unaltered by the stress concentration and the Code Case 1592 stress-to-rupture design curve shall be used.

TABLE 5.2-1

SUMMARY OF CODE, CODE CASES AND RDT
STANDARDS APPLICABLE TO DESIGN AND MANUFACTURE
OF REACTOR VESSEL, CLOSURE HEAD AND GUARD VESSEL

Component/Criteria	Reactor Vessel	Closure Head* Pressure Boundary	Closure Head* Internals (as appropriate)	Guard Vessel
Section III ASME Code, 1974 Edition	Addenda thru Winter '74	Addenda thru Winter '74	Addenda thru Winter '74	Addenda thru Summer '75
	Class 1	Class 1	Class 1	Class 2**
ASME Code Cases	1521-1,1592-2,1593-0,1594-1,1595-1, 1596-1 (1682,1690 Optional)	1682,1690	1521-1 1592-4,1593-1	1592-4,1593-1,1594-1 if elected by supplier 1521-1 & 1682
RDT Standards Mandatory	E8-18T, 2/75 E15-2NB-T, 11/74 Amend thru 1/75	E15-2NB-T, 11/74 Amend thru 6/75	E15-NB-T, 11/74 Amend thru 6/75	E15-NB-T, 11/74 Amend thru 6/76
	F2-2, 8/73 Amend thru 7/75	F2-2, 8/73 Amend thru 7/75	F2-2, 8/73 Amend thru 7/75	F2-2, 8/73 Amend thru 7/75
	F3-6T, 12/74	F3-6T, 12/74***	F9-4, 9/74	F3-6T, 10/75 With Amend. 1/75
	F6-5T, 8/74 Amend thru 2/75	F6-5T, 8/74 Amend thru 2/75		F6-5T, 8/74 Amend thru 11/75
	F7-3T, 11/74	F7-3T, 6/75		F7-3T, 6/75
	F9-4T, 9/74	M1-1T, 3/75 M1-2, 3/75 Amend thru 7/75		F9-4, 9/74

*For those reactor vessel and closure head components internal to the pressure boundary special purpose high cycle fatigue curves and creep fatigue damage rules have been developed as discussed in ~~Appendix 5.2A~~ Section 4.2.2.3.2.3

Delete

APPENDIX 5.2A

Modifications to the High Temperature Design Rules for Austenitic Stainless Steel.

Creep-Fatigue Evaluation

Creep-fatigue evaluations will be performed in accordance with the applicable criteria except as modified herein.

The creep-fatigue damage rules of Paragraph T-1400 of Code Case 1592 consider creep damage accumulations resulting from stresses which are clearly compressive to be equally as damaging as creep damage accumulations from tensile stresses. The damaging effects of compressive stresses in a high temperature environment are known to vary considerably from one material to another. Strain controlled fatigue test data of austenitic stainless steels (304 and 316 SS) consistently point to compressive residual stresses having little or no deleterious effect. There is also test evidence that suggests that when subjected to alternate hold periods in both tension and compression that hold in compression has a healing effect on the damage produced by the tensile hold. Based upon these data, the creep-fatigue damage rules are modified as described in subsequent paragraphs.

The effects of the presence of stress concentrations on stress rupture properties are known to vary considerably with the material, geometry of the stress concentration, magnitude of the stress level, the environment, and life. In the case of austenitic stainless steels, test data consistently points to stress concentrations having a less severe effect on stress rupture strength than predicted using the analytical apprcaches of 1592 and F9-4 criteria, and in the case of 316 SST, there is a consistent trend to significant notch strengthening for some types of geometries, particularly with a service environment and life at the upper limit of those in the UIS. The rules of RDT F9-4T and Code Case 1592 require comparing the peak stress to the Code strength which is based upon smooth specimen data. They do not recognize that peak stresses may have no adverse effect on stress rupture strength nor do they recognize that non-uniform stress states may alter the strength of the material. Based upon test data, the creep damage rules are modified as described in subsequent paragraphs to allow the use of a peak stress to rupture design curve.

Modifications to Creep-Fatigue Damage Rules

In cases where, in the service life of the component, all three principal stresses are clearly compressive during a hold period, the creep-fatigue evaluation shall be modified as described herein. If prerequisites for the use of the modified rule are not met for a portion of a component's life, the creep-fatigue rules of T-1400 of Code Case 1592 shall be used without modification for that portion of the component's life. The modified rule is described in Items (1) to (7), where (1) to (5) are prerequisite conditions,
58 and Item (7) is a final applicability criteria to be satisfied.

MEB-68. Large thermal stresses arise in the outer region of the perforated area of the steam generator tubesheet to the rim. Creep rupture damage combined with fatique due to relaxation of high residual stresses limits life of the component. The ASME Code does not provide acceptance criteria for the design of the perforated plates in elevated temperature service.

Response:

The resolution of this item consists of the following actions:

A. The Applicant is committed to perform Mechanical Properties Tests to verify and supplement ASME Code and RDT Standards methods and design information for assuring the structural adequacy of the steam generator. Prototype Steam Generator Tests will be run to verify certain performance characteristics. Hydraulic Test Model, Large Leak Tests, Few Tube Tests, DBN Tests (departure from nucleate boiling), Tube Support Wear Tests, Modular Steam Generator Tests, Single Tube Performance, Stability and Interaction Tests, Tube-to-Tubesheet Weld Tests, Scale Hydraulic Model Feature Tests, and Flow-Induced Vibration Tests will also be conducted. The tests are needed to confirm the structural adequacy of the tubes.

B. The Applicant will carry out a confirmatory program to confirm the adequacy of the methods and criteria used to ensure the structural adequacy of the tubesheet for its intended lifetime. The specific tasks involved are as follows:

1) Develop effective properties of perforated region for use in inelastic design analyses.

2) Evaluate effects of thermal gradients and equivalent material property variations on ligaments near periphery of perforated region.

3) Extended existing Appendix A-8000 Code methods for calculating the linearized membrane, shear and in-plane bending* stresses in the ligaments using the equivalent solid plate stresses. Include all of these nominal stresses in the comparison with allowable primary membrane plus bending, and primary plus secondary allowables.

4) Develop methods of evaluating local cyclic plastic strain concentration effects based on equivalent solid plate stresses for use in the fatigue evaluation.

5) Develop methods of evaluating local cyclic creep strain concentration effects based on equivalent solid plate stresses for use in the fatigue evaluation.

6) Evaluate elastic follow-up into outermost ligaments and
   (i) Reclassify portion of discontinuity stresses caused by pressure and mechanical loads as primary in accordance with the associated amount of elastic follow-up that occurs during thermal transients.
   (ii) Reclassify portion of thermal stresses as primary in accordance with the amount of elastic follow-up that occurs during thermal transients.

7) Develop ratcheting evaluation methods for outermost ligaments based on elastic equivalent solid plate stresses reclassified per Item 6 and including nominal membrane, shear and in-plane bending* stresses.

8) Develop creep rupture damage evaluation methods for outermost ligaments based on equivalent solid plate stresses. The effects of elastic follow-up will reduce the amount of stress relaxation and increase the creep rupture damage.

9) Perform detailed tube-to-tubesheet joint analysis for tubes in high radial thermal transient region at periphery of the perforated region including local thermal effects.

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*In-plane bending occurs on either side of minimum ligament section creating a "kinking" type of failure mechanism.

MEB Item 69: Applicant should review the current version of the MEB 3-1 (Revision 1) to assure that other documents used for specifying pipe break locations provide an equivalent level of conservatism.

Response: MEB 3-1, Rev. 1, is now being applied to CRBRP systems with the exception of the main PHTS and IHTS piping. The piping integrity analysis addresses the main piping. The enclosed change pages for PSAR Section 3.6 provide the appropriate references to BTP MEB 3-1 and to BTP ASB 3-1.

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

CRBRP systems and components important to safety will be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures or other events.

3.6.1 Systems In Which Pipe Breaks are Postulated

3.6.1.1 Systems Inside Containment

Spontaneous ruptures of heat transport system (HTS) and auxiliary sodium piping inside containment are not considered credible because of the high quality of the piping, operating temperature and pressure conditions for this piping, the inert environment provided for it, and the capability of the leak detection system to provide an early warning of any breach in the piping boundary. As a result, massive failures of this sodium HTS piping have not been included in the design bases for CRBRP systems inside containment. A four inch crack, which leads to 8 gpm PHTS leak rates, has been chosen as the design basis. A description of the analyses and test results to support this position are presented in the Piping Integrity Status Report (Reference 2, Section 1.6).

A similar detailed evaluation has not been performed for the sodium piping in the auxiliary liquid metal systems. The piping parameters (e.g. t/D ratio), service conditions (temperature, pressure, duty cycle), monitoring and inservice inspection techniques are similar to those for the heat transport system. Based on this, the maximum credible crack length is not larger than the 4 inch crack specified for the PHTS. The cell liner design is proceeding on the basis of containing the 4 inch crack with pressures and temperatures that are characteristic of the associated sodium system.

The chilled water system piping ASB in-containment is moderate energy piping according to the definition in BTP ~~APCSB~~ 3-1: Pipe leaks are postulated in the piping and mitigated by the features discussed in Section 9.7.3.

3.6.1.2 Systems Outside Containment

For systems outside of MEB 3-1 containment, the intent of the guidelines in ~~Appendix C to~~ Branch Technical Position ~~APCSB 3-1 (J. F. O'Leary letter 7/12/73)~~ will be used as a basis for leak evaluations. Where seamless pipe is used longitudinal
31 breaks will not be postulated if all stresses are below 0.8 (1.2 S_h + S_A). Separation and isolation of equipment by arrangement as shown in the figures in Section 1.2, atmosphere separation as described in Section 3A, and equipment enclosure are provided to protect safety systems and components required to shutdown the reactor safely. The high and moderate energy piping systems outside containment are listed in Tables 3.6-1 and 3.6-2, with the PSAR Section which discusses the system and the potential results of pipe leaks. Chapter
27 15.0 also contains analyses of postulated pipe leaks.

3.6.1.2.1 Water/Steam Systems

The definitions contained in Appendix A to the Branch Technical Positions are considered to be applicable to the water and steam piping outside containment. The following is a tabulation of the high and moderate energy systems together with a discussion of the design features which protect the essential systems necessary to shut the reactor down and to mitigate the consequences of a postulated pipe break.

3.6.1.2.1.1 Steam Generator Auxiliary Heat Removal System (SGAHRS)

The elements of this system are described in Section 5.6.1.

ASB 3-1

The piping from the auxiliary feedwater isolation valves to the steam drum, the piping between the steam drum and the Protected Air Cooled Condenser (PACC), and the piping from the steam drum to the Auxiliary Feedwater Pump (AFWP) turbine drive isolation valve are high pressure as defined in Appendix A to ~~the~~ a BTP, and will be evaluated for postulated ruptures. Because these pipes are located outside of the cells containing the major auxiliary feedwater components, a continued supply of auxiliary feedwater will be available after a postulated rupture. Separation of the HTS loops and their respective cells and Steam Generator Building flooding protection
45 provisions (Section 7.6.5) prevent propagation of a pipe rupture event to adjacent loops and thus the essential systems to mitigate the consequences of the rupture are maintained.

The piping runs from the AFWP to the AFW isolation valves and from the turbine drive steam supply isolation valve to the turbine drive are low-pressure and low-temperature lines during normal plant operation. Both lines are subjected to high-pressure during AFW operating periods and the turbine supply line is also subjected to high temperature conditions during the time the turbine is operational. However, the SGAHRS operating time is anticipated to be less than 2% of the plant operating time since the auxiliary feedwater portion of SGAHRS will not be utilized unless the Normal Heat Rejection ystem (main condenser) or Feedwater Supply System has been lost. Therefore, this piping will be evaluated as moderate energy piping for through the wall cracks during normal plant operations. The primary concern for a crack in this piping is for protection of the major auxiliary feedwater components from the accumulated water that has leaked. The major components are elevated to provide this protection and to prevent the propagation of event consequences. Other essential systems for reactor shutdown are not impacted by low temperature and low pressure leaks from this piping.

No piping breaks will be postulated for the low pressure and low
27 temperature piping run from the Protected Water Storage Tank to the AFWP.

3.6.1.2.1.2 Steam Generator System

ASB

The elements of this system are described in Section 5.5. The majority of piping in this system is high pressure as defined in Appendix A of Branch Technical Position ~~APCSB~~ 3-1. The feedwater piping between the FW control valve and the steam drum, the recirculation and main steam piping joining the evaporators and steam drum, and the steam piping from the steam drum to superheated steam isolation valves will be evaluated for postulated ruptures. Because this equipment is separated by loop into building cells and Steam Generator Building
45 flooding protection is provided (Section 7.6.5), the effects of these ruptures will not propagate to adjacent cells and thus essential systems for non-ruptured HTS loops necessary to mitigate the rupture consequences will remain operable.

3.6.1.2.1.3 Main Steam, Condensate and Feedwater Systems

The elements of these systems are discussed in Chapter 10.0. These systems are located in the Turbine Generator Building. Safety related equipment in the Steam Generator Building is protected by the hardened wall of the Steam Generator Building.

49 3.6.1.2.1.4 Chilled and Treated Water Systems

The elements of these systems are discussed in Sections 9.7 and 9.9. They are classified as moderate energy systems. The primary protection against pipe leaks in those systems are floor drains, isolation provisions and separation of redundant equipment. Separation of equipment by loops assures operation of redundant loops of essential systems
27 in the event of a piping leak.

3.6.1.2.2 Sodium and NaK Systems

Sodium and NaK are unique relative to water in that their boiling temperatures are significantly higher than their normal operating temperatures in CRBRP applications. Therefore, in comparison with conventional water systems, no CRBRP sodium or NaK systems operate with any significant amount of internal fluid stored energy. The highest temperature sodium system outside the CRBRP containment operates at 965°F vs. the sodium saturation temperature of 1630°F at atmospheric pressure. Typical NaK operating temperatures are in the range of 500°F vs. the NaK saturation temperature of ~1450°F at atmospheric pressure. Therefore, from the standpoint of fluid stored energy, the CRBRP sodium and NaK systems outside containment contain essentially no stored energy and are therefore moderate energy systems.

In addition to fluid considerations, total system operating modes have been evaluated relative to their impacts upon failure. In spite of fluid conditions, systems can have high internal energy, e.g., subcooled liquid systems with saturated liquid pressurizers. The operating pressure of essentially all of the CRBRP systems is solely dependent on the developed discharge head of the system pump. This energy source is generated only to accommodate the resistance of the system to fluid flow. When considering system failures, this resistance to flow immediately diminishes resulting in immediate loss of system pressure. There is therefore, very little energy available in a failed system to induce large reaction forces at the failure point. The initial reaction forces diminish within milliseconds following a large failure.

The only sodium or NaK system within CRBRP which has an external pressure source is the intermediate heat transport system which utilizes an expansion tank that contains pressurized cover gas. Even though this system does contain a pressure source, the total system pressure is well below 275 psig (~225 psig).

ASB

From the foregoing discussion, it is concluded that sodium and NaK systems are moderate energy systems within the intent of the definitions given in Appendix A of Branch Technical Position ~~APCSB~~ 3-1. They will all be evaluated in accordance with the intent of the guidelines ~~of the J. F. O'Leary letter of 7/12/73~~. The following is a tabulation of these systems and a discussion of the design features which protect the systems necessary to shut the reactor down and to mitigate the consequences of a postulated pipe leak. BTP MEB 3

3.6.1.2.2.1 Intermediate Heat Transport System (Outside Containment)

BTP MEB 3-1

The intermediate heat transport system piping outside containment will be evaluated in accordance with the intent of the requirements for moderate energy piping established by ~~the J.F. O'Leary letter of 7/12/73~~. The primary protection provided for postulated leaks in this piping is separation of the
27 IHTS loops by building cells. Sodium catch pans are included in the design

3.6.2 Pipe Break Criteria

BTP MEB 3-1

For sodium and NaK lines, the approach and criteria described in Section 3.6 will be followed. For piping of other fluid systems inside the containment, the intent of ~~Regulatory Guide 1.46~~ will be used as a guide for postulating break locations, break sizes and orientations; and pipe restraints will be provided as necessary.

There is no high-pressure water and steam piping inside the containment of the CRBRP. For water/steam piping outside the containment, pipe breaks will be postulated in accordance with the bases described in Sections 3.6.1.1 and 3.6.1.2. For those piping runs for which pipe breaks are to be postulated, the criteria as set forth in ~~Appendix A to J. F. O'Leary letter of 7/12/73~~ BTP MEB 3-1 will be followed with respect to: pipe break locations, break sizes and orientations. However, since the SGS and SGAHRS steam/water piping is of seamless construction and longitudinal pipe welds are not used
29 at component connections, longitudinal pipe breaks will only be postulated in nominal pipe sizes of 4 inches and larger at terminal ends and intermediate locations where the stresses exceed 0.8 (1.2 S_h + S_A) based on the loadings specified in ~~Appendix A to the O'Leary letter~~, unless the axial stress is greater than 1.5 times the circumferential stress.

BTP MEB 3-1.

The essence of steam/water pipe break criteria is that for a water/steam pipe break in any loop, including the steam/water release, the effects must be restricted to that loop and must not impair operability of the other two loops, SGAHRS equipment in the other two loops, or SGAHRS equipment in the lower cells of the auxiliary bay. The effects of a water/steam pipe break shall not lead to an uncontrolled sodium-water reaction and result in hydrogen gas release into a cell of the affected loop unless the consequences are shown to be within acceptable limits. Protection of critical components must be provided by (1) physical separation, (2) pipe whip restraints, or (3) impingement shields. This protection will consider pipe movement caused by the reaction force of the jet and impingement of steam/water on adjacent components and piping.

3.6.3 Design Loading Combinations

In all locations where piping breaks are postulated to occur, analysis will be performed on the unrestrained piping system, except where it has been shown that pipe dynamics resulting from a pipe break can be neglected. This analysis is required to insure against possible damage to neighboring reactor coolant boundary components and all essential equipment located within containment and the SG cells. Consideration of damage propagation to adjacent piping will be given as appropriate, consistent with the intent of Regulatory Guide 1.46. For the pipe supports and structures, the load combination as defined in Section 3.8.3.3.10 will be used.

For jet impingement interactions, piping subject to jet impingement shall be assumed to fail if: a) the jet impingement load exceeds the load carrying capacities of the pipe support system; or b) the pipe stress exceeds the allowables for the faulted loading condition as defined for the applicable ASME-III classifications.